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JUN 0 5 2018

10 CFR 50.73

Serial: RA-18-0037

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Brunswick Steam Electric Plant, Unit No. 1
Renewed Facility Operating License No. DPR-71
Docket No. 50-325
Licensee Event Report 1-2018-002

In accordance with the Code of Federal Regulations, Title 10, Part 50.73, Duke Energy Progress, LLC, submits the enclosed Licensee Event Report (LER). This report fulfills the requirement for a written report within sixty (60) days of a reportable occurrence.

This document contains no regulatory commitments.

Please refer any questions regarding this submittal to Mr. Lee Grzeck, Manager – Regulatory Affairs, at (910) 832-2487.

Sincerely,

William R. Gideon

MAT/mat

Enclosure: Licensee Event Report

cc (with enclosure):

U. S. Nuclear Regulatory Commission, Region II
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NRC FORM 366 (02-2018)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

(See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

1. Facility Name	2. Docket Number	3. Page
Brunswick Steam Electric Plant (BSEP), Unit 1	05000325	1 OF 4

4. Title

Automatic Reactor Trip during Stator Coolant System Realignment

5. Event Date			6. LER Number			7. Report Date			8. Other Facilities Involved	
Month	Day	Year	Year	Sequential Number	Rev No.	Month	Day	Year	Facility Name	Docket Number
04	07	2018	2018	- 002	- 00	06	05	2018	Facility Name	05000
									Facility Name	05000

9. Operating Mode: 1

10. Power Level: 100

11. This Report is Submitted Pursuant to the Requirements of 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(ii)
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(iii)
	☐ 50.73(a)(2)(i)(C)	☐ Other (Specify in Abstract below or in NRC Form 366A)	

12. Licensee Contact for this LER

Licensee Contact	Telephone Number (Include Area Code)
Lee Grzeck, Manager – Regulatory Affairs	(910) 832-2487

13. Complete One Line for each Component Failure Described in this Report

Cause	System	Component	Manufacturer	Reportable to ICES	Cause	System	Component	Manufacturer	Reportable to ICES

14. Supplemental Report Expected	15. Expected Submission Date	Month	Day	Year
☐ Yes (If yes, complete 15. Expected Submission Date) ☒ No				

Abstract (Limit to 1400 spaces, i.e., approximately 14 single-spaced typewritten lines)

On April 7, 2018, at 0836 Eastern Daylight Time (EDT), Unit 1 automatically tripped during realignment of the Stator Cooling System. The trip was uncomplicated with all systems responding normally. Reactor water level reached low level 1 (LL1) due to the reactor trip. Per design, the LL1 signal resulted in Group 2, Group 6, and Group 8 Primary Containment Isolation System (PCIS) actuations. The Reactor Protection System (RPS) and PCIS actuations are being reported in accordance with 10 CFR 50.73(a)(2)(iv)(A) as an event or condition that resulted in manual or automatic actuation of systems listed in 10 CFR 50.73(a)(2)(iv)(B).

The direct cause of this event was a design deficiency which resulted in an unanticipated Turbine Protection System actuation that caused a main turbine trip and subsequent automatic reactor trip. The root cause of the design deficiency was determined to be human performance related. Corrective actions include training and procedure revisions.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Brunswick Steam Electric Plant (BSEP), Unit 1	05000325	2018	- 002	- 00

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Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

Background

Initial Conditions

At the time of the event, Unit 1 was in Mode 1 (i.e., Power Operation), at approximately 100 percent of rated thermal power (RTP). No safety-related equipment was inoperable at the time of the event.

Reportability Criteria

The Reactor Protection System (RPS) [JC] and Primary Containment Isolation System (PCIS) [JM] actuations are being reported in accordance with 10 CFR 50.73(a)(2)(iv)(A) as an event or condition that resulted in manual or automatic actuation of systems listed in 10 CFR 50.73(a)(2)(iv)(B). The event was initially reported to the NRC on April 7, 2018 (i.e., Event Number 53319).

Event Description

On April 7, 2018, realignment of the Unit 1 Stator Cooling System [TJ] was being performed in accordance with plant procedure 0PT-36.1, Stator Cooling System. On a monthly basis, the operating Stator Cooling pump is alternated. This is accomplished by starting the reserve pump, followed by shutting down the operating pump. When the reserve pump (i.e., the 1B pump) was started, a Loss of Stator Cooling and a Stator Coolant Low Flow alarm were received in the control room. Both Stator Cooling pumps were in service and reports from the field indicated normal Stator Cooling System pressure. Computer data indicated the expected increase in Stator Cooling System flow when the second pump was started. The Control Room staff evaluated indications for the Stator Cooling System and found no evidence that there were any problems with the system or impact on Stator temperature. The staff continued with restoration steps for 0PT-36.1. However, coincident with the alarms, a 120 second timer started; at the completion of which an automatic main turbine trip occurred. This resulted in an automatic reactor trip at 0836 Eastern Daylight Time (EDT).

The scram was uncomplicated with all systems responding normally. Reactor water level reached low level 1 (LL1) due to the reactor trip. Per design, the LL1 signal resulted in Group 2 (i.e., floor and equipment drain isolation valves), Group 6 (i.e., monitoring and sampling isolation valves) and Group 8 (i.e., shutdown cooling isolation valves) isolations. The LL1 isolations occurred as designed; the Group 8 valves were closed at the time of the event.

Event Cause

The direct cause of this event was a design deficiency which resulted in an unanticipated Turbine Protection System [JJ] actuation that caused a main turbine trip and subsequent automatic reactor trip. This was the first performance of 0PT-36.1 after the recently completed Unit 1 refueling outage during which a new Turbine Control System (TCS) [JJ] was installed.

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The original Stator Cooling System measured flow, pressure, and temperature and provided input to the Electro-Hydraulic Control (EHC) Turbine Protection System to initiate a main turbine trip on low flow, low pressure, or high temperature. For flow, the Stator Cooling System had a single flow differential pressure switch, which represented a single point scram vulnerability. To address the single point vulnerability, the flow differential pressure switch was replaced with three differential pressure transmitters, which input to a 2-out-of-3 trip logic. The differential pressure transmitters have a range of 0 to 600 gpm. The new TCS logic is designed to detect faulted instruments and initiate a main turbine trip if all the instruments for the protective function are faulted. The high out-of-range faulted instrument setpoint for the differential pressure transmitters corresponds to 612 gpm. Normal Stator Cooling System flow is approximately 550 gpm to 570 gpm with one pump in service but routinely exceeds 612 gpm with two pumps operating. When the reserve Stator Cooling pump was started during the performance of procedure 0PT-36.1, system flow exceeded 612 gpm and the subsequent main turbine and reactor trips occurred.

The root cause of the design deficiency was determined to be human performance related. Critical thinking tools (i.e., procedure prompts, operating experience application, risk identification, and supervisory oversight) were not effectively used to assure successful implementation of the new TCS. This resulted in the following shortcomings: (1) design documentation did not include sufficient information to adequately assess the impact and risk of the new faulted instrument trip feature such that reviewers could identify the vulnerability that was introduced; (2) design documentation did not include an assumption that the span of instrumentation bounded operational conditions that would be expected to be experienced, which lead to an inadequate evaluation of the high out-of-range faulted instrument setpoint selection and assumptions; and (3) operating experience was not effectively used to ensure that all installed instrument spans were identified for appropriate adjustments.

Safety Assessment

The safety significance of this event is minimal. The automatic reactor trip was not complicated and all safety-related systems operated as designed.

Corrective Actions

The following corrective actions were completed.

- A software change was implemented to remove the automatic bypass and trip input of faulted high-out-of-range Stator Cooling Flow differential pressure transmitters.
- The TCS modification was reviewed to verify that process inputs for trip functions in TCS had adequate margin to ensure faulted instrument trip setpoints would not be exceeded during postulated operation including transients.

The following corrective actions are currently planned. Any changes to the corrective actions and schedules noted below will be made in accordance with the site's corrective action program.

- Training will be conducted to present a case study of lessons learned from failed barriers for design development (i.e., use of procedure prompts, operating experience application, risk identification, and

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supervisory oversight) to identification, documentation, and verification of assumptions to appropriate Engineering personnel. The case study will be incorporated into backbone Design Control recurring training. This action is currently scheduled to be completed by October 31, 2018.

- Procedure AD-EG-ALL-1157, *Engineering of Plant Digital Systems and Components*, will be revised to provide more specific guidance on evaluation of response to out-of-range or unexpected values to consider margin to achievable plant conditions. This action is currently scheduled to be completed by December 20, 2018.

- Procedure AD-EG-ALL-1132, *Preparation and Control of Design Change Engineering Changes*, will be revised to add consideration of new or modified functions that will cause a reactor trip or Engineered Safety Feature (ESF) actuation to ensure that methods of initiating the trip or ESF actuation are evaluated for unintended actuation. This action is currently scheduled to be completed by December 20, 2018.

Previous Similar Events

A review of LERs for the past three years identified the following previous similar event where a modification introduced an unforeseen consequence.

- LER 1-2016-002, Revision 1, dated August 8, 2016, reported a condition that could have prevented the fulfillment of a safety function and operation prohibited by the Technical Specifications due to inoperability of Emergency Diesel Generator (EDG) 3. The root cause of the EDG 3 inoperability was a design vulnerability, associated with relaxation of the EDG 3 fuse holder fingers, which was not properly mitigated. The corrective actions associated with LER-2016-002, Revision 1, were focused on correcting the design vulnerability and the use of fuse block devices. These actions could not have reasonably been expected to prevent the condition reported in LER 1-2018-002.

Commitments

No regulatory commitments are contained in this report.